NON-PUBLIC?: N
ACCESSION #: 9312210406
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Surry Power Station, Unit 2 PAGE: 1 OF 7

DOCKET NUMBER: 05000281

TITLE: Unit 2 Automatic Reactor Trip Due to Low Steam Generator
Level in Coincidence With Feed Flow Mismatch Following
Closure of All Three Main Feedwater Regulating Valves
EVENT DATE: 11/15/93 LER #: 93-006-00 REPORT DATE: 12/13/93

OTHER FACILITIES INVOLVED:
DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 95%

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M. R. Kansler, Station Manager TELEPHONE: (804) 357-3184

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: EI COMPONENT: BKR MANUFACTURER: S345
REPORTABLE NPRDS: Yes

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On November 15, 1993, at 2019 hours, with Unit 1 at 95% power and Unit 2 at 95% power, Unit 2 experienced an automatic reactor trip. The trip occurred when an electrical circuit failure caused all three Main Feedwater Regulating Valves to go shut, isolating feedwater flow to all three steam generators. The resultant feedwater flow/steam flow mismatch coincident with low steam generator water levels actuated the Reactor Protection System (RPS), and the reactor trip breakers opened. The RPS functioned as designed, and post-trip response was satisfactory. The reactor was placed in a safe, hot shutdown condition, and the health and safety of the public were not affected. The electrical failure was traced to a defective circuit breaker which was replaced. Three similar breakers whose failure alone could cause a reactor trip were also replaced. Additional corrective actions are under evaluation. This

report is required pursuant to 10CFR50.73(a)(2)(iv).

END OF ABSTRACT

Figure "REQUIRED NUMBER OF DIGITS/CHARACTERS FOR EACH BLOCK" omitted.

1.0 DESCRIPTION OF THE EVENT

On November 15, 1993, at 2019 hours, Unit 2 experienced an automatic reactor trip from 95% power. An electrical circuit failure caused all three Main Feedwater Regulating Valves (MFRV), 2-FW-FCV-2478, 2488, 2498, (EIIS-JB, CV) to shut, isolating feedwater flow to the steam generators. The reactor tripped as a result of low steam generator level in coincidence with a feedwater flow/steam flow mismatch in the "C" Steam Generator (SG), 2-RC-E-1C (EIIS-AB,SF). The loss of feedwater flow and decreasing levels in all three generators had been noted by Control Room Operators who were attempting to correct the condition when the trip occurred. The turbine (EIIS-TA) and main generator (EIIS-TB) tripped as designed. The Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) also actuated as designed. The Auxiliary Feedwater (AFW) Pumps, 2-FW-P-2, 3A, and 3B (EIIS-BA, P), automatically started and supplied feedwater to the steam generators as designed.

Control Room Operators responded to the trip in accordance with emergency and other operating procedures. Plant response was as expected except for the following:

o Reactor Coolant System (RCS) (EIIS-AB) temperature decreased to about 525 degrees F, recovering to the design value of 547 degrees F after Control Room Operators shut the Main Steam Trip Valves (MSTV), 2-MS-TV-201A,B, and C (EIIS-SB,ISV).

o Individual Rod Position Indication (IRPI) md bottom light (EIIS-AA, IL) for Control Rod M-10 was slow in illuminating. (The rod bottom light illuminated at approximately 2108 hours.)

o Thirteen minutes after Motor Driven AFW Pump 2-FW-P-3A automatically started, smoke was detected at the inboard packing gland. The pump was secured and declared inoperable as a precautionary measure. The low level condition on all three Steam Generators had cleared several minutes earlier.

TEXT PAGE 3 OF 7

A four-hour non-emergency report was made to the Nuclear Regulatory Commission in accordance with 10CFR50.72(b)(2)(ii) at 2344 hours. This event is being reported pursuant to 10CFR50.73(a)(2)(iv) as an automatic actuation of the Reactor Protection System (RPS) (EIIS-JC).

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

When the reactor trip occurred, RPS actuations functioned as designed, and all control rods inserted into the core. The electrical buses transferred properly and off-site power was maintained throughout the event. The emergency diesel generators remained operable in automatic but were not required to start. The Auxiliary Feedwater Pumps staKed and supplied feedwater to the steam generators as designed. Station operating personnel acted promptly to place the plant in a stable, hot shutdown condition. The shutdown margin of reactivity was calculated and found to be satisfactory. No conditions adverse to safety resulted from this event, and the health and safety of the public were not affected.

3.0 CAUSE OF THE EVENT

The reactor tripped as designed when the "C" Steam Generator experienced a feedwater flow/steam flow mismatch coincident with a low steam generator water level. The cause of the loss of feedwater flow and resultant loss of water level was the closure of all three MFRVs. The loss of flow and decreasing water levels had been noted in all three steam generators, and Control Room Operators were attempting to restore normal operating conditions when the trip occurred. The MFRVs are air actuated valves whose air is supplied via two redundant trains of solenoid operated trip valves. The MFRVs went shut unexpectedly when the circuit breaker supplying DC power to one train of the solenoid operated trip valves opened. During certain off-normal plant conditions, these solenoid operated valves interrupt the air supply to the MFRV actuators, causing the valves to shut and isolate the feedwater supply. Because of their "fail safe" design, loss of power to either train of the

TEXT PAGE 4 OF 7

solenoid operated valves produces the same result as an actuating signal, venting the air from the MFRV actuators and causing the valves to shut.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Operators acted promptly to place the plant in a safe, shutdown condition in accordance with emergency and other operating procedures. The Shift Technical Advisor monitored the critical safety function status trees to verify that unit conditions were acceptable. A multi-discipline Root Cause Evaluation Team was formed to investigate the event.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

o MFRV Circuitry

- Electrical checks and visual inspection of the circuitry and components involved revealed no abnormal condition that could have contributed to the tripping of the DC supply breaker (Breaker 16 in Distribution Panel DC 2-1). No evidence of water damage, steam damage, or electrical arcing was found. Limit switch and terminal covers were in place and tightly sealed.

- DC Breaker 16 was discovered in the trip free condition and would not reset. The breaker was returned to the vendor for further testing which revealed the following:

-- A crack had developed near the location where the calibration screw engages the terminal side of the breaker.

-- This crack could have increased the electrical resistance of the terminal thereby changing its calibrated set point and causing the breaker to trip at a different current value.

-- Because of resultant mechanical deformation, the crack also prevented the resetting of the breaker.

- Breaker 16 was replaced.

o RCS Temperature Decrease

- Cooldown of the RCS to temperatures below the design value following reactor trips has been a recurring problem.

Inadequate isolation of steam loads, leaking valves, and improper valve line ups have been identified and corrected in the past; however cooldown problems continue to occur. Although cooldowns have not exceeded design basis limitations, further investigation into the cooldown problem is being conducted. The evaluation will principally focus on the design, operation, response, and maintenance of the steam dump valves and their associated control program. Appropriate corrective actions will be taken based on the results of this evaluation.

o IRPI rod bottom light for Control Rod M-10

- Control Rod M-10 IRPI has exhibited slow response following reactor trips for several years. This condition has been attributed to residual permeability in the control rod drive mechanism housing.

- A hot rod drop test conducted subsequent to the reactor trip verified that Control Rod M-10 is fully operable.

- An engineering evaluation of Control Rod M-10 has been performed. A number of recommendations have been implemented, and additional items are scheduled for future outages.

o Motor Driven AFW Pump 2-FW-P-3A

- During the last Unit 2 Refueling Outage, the packing of all three AFW Pumps was replaced with material different from that which had been used in the past. This braided graphite foil packing is

manufactured by Slade, Inc., and is expected to smoke upon initial pump run-in as the acrylic resin used in the manuf
cturing process "bakes out" of the packing material.
The last documented instance of smoke having been detected on AFW Pump 2-FW-P-3A was on October 29, 1993, during monthly surveillance testing. The pump was repacked, some adjustments were made, and the pump was tested satisfactorily and returned to service.

- During the pump's run on November 15, smoking at the packing was again detected. Since a certain period of pump operation is required to bake out the resin, this condition could have resulted from additional curing after the return-to-service run on October 29. Nevertheless, the pump was secured and declared inoperable as a precautionary measure.

- Also as a precautionary measure, the packing on all three Unit 2 pumps was replaced with the originally installed style of packing (Garlock). The Unit 1 pumps still have this original packing material installed.

6.0 ACTIONS TO PREVENT RECURRENCE

As a result of a Root Cause Evaluation conducted subsequent to the reactor trip, the following actions have been taken or directed:

- Three other breakers whose failure alone could result in a reactor trip have been replaced as a precautionary measure.

- The corresponding Unit 1 breakers will be inspected and replaced as required during the next outage of sufficient length.

- An evaluation is underway to determine if additional corrective or preventive maintenance actions should be pursued with regard to this type of circuit breaker. Appropriate corrective actions will be taken based on the results of this evaluation.

7.0 SIMILAR EVENTS

LER S2-90-003 Manual Reactor Trip Due to Failure of "A " Main Feedwater Regulating Valve (blockage of positioner air supply inlet filter/orifice assembly).

LER S2-90-004 Manual Reactor Trip Following Inadvertent Grounding of the "A" Main Feedwater Regulating Valve Control Signal During Testing.

LER S2-93-003 Automatic Reactor Trip Due to Low Steam Generator Level Coincident With Steam/Feedwater Flow mismatch Resulting From Spurious Closure of "A" MFRV.

8.0 ADDITIONAL INFORMATION

Failed Component:

Square D Molded Case Circuit Breaker Model A1B-215

ATTACHMENT TO 9312210406 PAGE 1 OF 1

10CFR50.73

Virginia Electric and Power Company
Surry Power Station
P. O. Box 315
Surry, Virginia 23883

December 13, 1993

U. S. Nuclear Regulatory Commission Serial No.: 93-772
Document Control Desk SPS:VAS
Washington, D. C. 20555 Docket No.: 50-281
License No.: DPR-37

Dear Sirs:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 2.

REPORT NUMBER

50-281/93-006-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

M. R. Kansler
Station Manager

Enclosure

cc: Regional Administrator
101 Marietta Street, NW, Suite 2900

Atlanta, Georgia 30323

M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

*** END OF DOCUMENT ***